NON-PUBLIC?: N
ACCESSION #: 900910049
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK - UNIT 1 PAGE: 1 OF 8

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP ON LOSS OF FEEDWATER FLOW CAUSED BY A LOOSE CONTROL
POWER FUSE
EVENT DATE: 08/08/90 LER #: 90-023-00 REPORT DATE: 09/05/90

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 017

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: G.P. McGEE TELEPHONE: (817) 897-5477
SUPERVISOR, COMPLIANCE

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SJ COMPONENT: FUB MANUFACTURER: X999
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On August 8,1990, Comanche Peak Steam Electric Station Unit 1 was in Mode 1, Power Operations, with reactor power at 17 percent. A loose fuse in the Main Feedwater control power circuit caused closure of a valve in the feedwater flow path to Steam Generator number 4. Water level in Steam Generator number 4 decreased to the Lo-Lo level setpoint, initiating a reactor trip signal. Corrective actions included inspection of similar components in other applications, maintenance on the malfunctioning component, and personnel training.

END OF ABSTRACT

TEXT PAGE 2 OF 8

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

An event or condition that resulted in the manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

B. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On August 8, 1990, at 0508 CDT, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operations, with reactor power at 17%.

C. STATUS OF STRUCTURES,SYSTEMS,OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE
TIMES

On August 8, 1990, just prior to the event, Control room personnel were stabilizing reactor and turbine generator power following synchronization with the grid. The steam generators (EIIS:(SG)(SB)) were being supplied with main feedwater (EIIS:(SJ)) through the Feedwater Preheater Bypass Valve (FPBV) (EIIS:(SJ)(ISV)) and the Feedwater Regulating Bypass Valve (FRBV) (EIIS:(SJ)(FCV)) (refer to Figure 1).

Figure 1 omitted.

At approximately 0500 CDT Control Room personnel were responding to a High Auxiliary Feedwater (EIIS:(BA))temperature alarm. In accordance with operating procedures, the Auxiliary Feedwater flow control valves (EIIS:(BA)(FCV)) were closed and both motor driven Auxiliary Feedwater Pumps (EIIS:(BA)(P)) were started. At approximately 0504 CDT the FPBV failed closed, and at 0505 the Steam Generator number 4 Lo Level alarm annunciated in the Control Room (EIIS:(NA)). The Balance of Plant Reactor Operator (utility, licensed) responded by verifying that the

FRBV was in automatic and by taking the controller to the full open demand position. The operator then increased the speed of the operating main feedwater pump (EIIS:(SJ)(P)) and throttled feedwater flow to Steam Generators 1, 2, and 3. Operating personnel speculated that the cooler Auxiliary Feedwater was leaking past the Auxiliary Feedwater flow control valve (EIIS:(BA)(P)) causing the decrease in Steam Generator level. The Auxiliary Feedwater containment isolation valve (EIIS:(BA)(ISV)) was closed to stop the suspected steam generator shrink.

Operating personnel observed flow to Steam Generator number 4 on one of two feedwater flow instruments (EIIS:(SJ)(Fl)); the other flow instrument had been taken out of service earlier. Initiation of a manual Reactor trip was discussed, but supervisory personnel (utility, licensed) felt that level recovery was possible based on level trend and indicated flow. At 0508 the Reactor tripped on Lo Lo level in Steam Generator number 4. Operating personnel responded in accordance with emergency operating procedures, stabilizing the plant in Mode 3. At 0648 the NRC was notified of the event via the Emergency Notification System line in accordance with 10CFR50.72.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR
PROCEDURAL OR PERSONNEL ERROR

Immediately following the event a review of ERF computer (EIIS:(CPU)(ID)) data revealed that feedwater to Steam Generator number 4 had isolated prior to the event. A work order was initiated to determine the equipment malfunction which caused the FPBV closure. During troubleshooting activities, control power was restored when the termination cabinet door was opened. Examination revealed that a control power fuse (EIIS:(SJ)(FU)) was apparently loose in the holder (EIIS:(SJ)(FUB)). Troubleshooting activities concluded that the loose fuse resulted in a loss of power to the Train A solenoid operated air supply valve (EIIS:(PSV)). The valve must be energized for the FPBV to open. When the fuse was reinstalled, control power was restored and the FPBV functioned normally.

II. COMPONENT OR SYSTEM FAILURES

A. FAILED COMPONENT INFORMATION

Buchanan Construction model 361 fuse block

B. FAILURE MODE,MECHANISM AND EFFECT OF EACH FAILED COMPONENT

Water level in Steam Generator number 4 decreased to the Lo Lo setpoint when Feedwater flow was lost following closure of the Feedwater Preheater Bypass Valve.

C. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

A loose fuse in the FPBV control power circuit caused a loss of control power to the Train
A solenoid operated air supply valve, isolating instrument air (EIIS:(LD)) to the FPBV and causing it to close.

D. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - no failures of components with multiple functions have been identified.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The Reactor Protection System (EIIS:(JC)) and Auxiliary Feedwater System actuated during the event; all associated components within these systems functioned as designed.

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable - there were no safety systems which were rendered inoperable due to a failure.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

A loss of normal feedwater resulting from pump failure, valve malfunction, or loss of offsite power leads to a reduction in the capability of the secondary system to remove heat generated in the reactor core. These events are analyzed in section 15.2.7 of the CPSES Final Safety Analysis Response (FSAR) which uses conservative assumptions in the analysis to minimize the energy removal capability of the Auxiliary Feedwater system. The reactor trip on August 8 occurred at 17 percent reactor power and all systems and components functioned as designed. The event is completely bounded by the FSAR accident analysis which assumes an initial power level of 102 percent and the worst single failure in the Auxiliary Feedwater system. The event could have occurred at a maximum power level of approximately 20 percent; above this point feedwater flow is realigned through the FRVs, and failure of the FPBV would not result in a loss of feedwater flow. At this increased power level and with the worst case single Auxiliary Feedwater system failure, the event continues to be bounded by the FSAR accident analysis. At full power operations a similar failure of a different valve attributable to the same cause would still be bounded by the FSAR accident analysis. It is concluded that the event of August 8 did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

IMMEDIATE CAUSE

The immediate cause of the reactor trip was Lo-Lo level in Steam Generator number 4 resulting from a loss of feedwater flow following closure of the Feedwater Preheater Bypass Valve. The valve failed closed as designed on a loss of control power due to a loose fuse in the control power circuit.

ROOT CAUSE

Engineering evaluation was performed in an attempt to identify possible causes of the fuse clip failure. A number of possibilities were examined; however, no conclusive evidence could be found to support any hypothesis. The most probable cause of the fuse clip failure is a combination of cycling of the clip during the extended plant construction/testing phase leading to fatigue of the clip tangs, and thermal effects on the tangs due to loss of contact surface area. Repeated removal and reinsertion of fuses results in a relaxation of the tang material and a consequent decrease in tension applied to the fuse. This can in turn lead to a reduction in contact surface area between the fuse and the clip and an increase in electrical resistance. The resultant thermal effects on the tang are ill defined and difficult to predict but lead in general to minor changes in tang geometry which can in turn further alter the contact surface areas. It is probable that this thermal cycling led eventually to a decrease in current carrying capacity across the contact surface sufficient to deenergize the affected solenoid operated control valve.

V. CORRECTIVE ACTIONS

A. IMMEDIATE

The immediate response of Control Room personnel was directed toward plant recovery following the event. The Instrument and Control Department initiated troubleshooting activities in cooperation with operations personnel to identify the cause of the valve closure. Corrective maintenance was performed on the malfunctioning fuse clip. The Steam Generator number 4 feedwater flow indicator was determined to be sticking at the lower end of its range; maintenance and calibration was performed on the instrument. Management responded by initiating incident investigation activities and engineering evaluation to address issues identified following the event.

B. ACTIONS TO PREVENT RECURRENCE

Prior to reentry into Mode 2, an inspection was performed on a selected population of similar fuse holders in safety related systems which assured that loose fuses do not represent a more generalized problem. The fuse clip will be replaced during a future maintenance opportunity.

In order to alert personnel of the potential for the condition, a Lessons Learned memo has been generated instructing Operations and Maintenance personnel to inspect fuse holders when installing fuses to ensure good working order. The lessons learned from this event will be incorporated into operator training.

VI. PREVIOUS SIMILAR EVENTS

There have been no previous Reactor trips or Engineered Safety
Features actuations attributable to loose fuses.

ATTACHMENT 1 TO 9009100049 PAGE 1 OF 1

Log # TXX-90292
File # 10200
907.3
910.4
915.2
Ref. # 50.73(a)(2)(iv)

William J. Cahill, Jr.
Executive Vice President September 4, 1990

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY
FEATURE
LICENSEE EVENT REPORT 90-023-00

Gentlemen:

Enclosed is Licensee Event Report 90-023-00 for Comanche Peak Steam
Electric Station Unit 1, "Reactor Trip on Loss of Feedwater Flow Caused
by a Loose Control Power Fuse."

Sincerely,

William J. Cahill, Jr.

TAH/daj

Enclosure

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

400 North Olive Street L.B. 81 Dallas,Texas 75201

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